



Emergency Management Competency 1.3

Competency 1.3 Emergency management personnel shall demonstrate a working level knowledge of health physics and radiation protection to oversee emergency activities and provide guidance in mitigating emergencies.

1. SUPPORTING KNOWLEDGE AND/OR SKILLS

- a. Describe the different types of radiation.
- b. Discuss the fundamentals of radiation protection as related to emergency response.
- c. Describe the relationship between dose and radiological injury.
- d. Discuss the following terms and concepts: bioaccumulation, biological half-life, intake, contamination, exposure, and criticality.
- e. Describe the types, uses, and limitations of radiation detecting and monitoring equipment.
- f. Discuss the emergency procedures associated with radiological releases into the environment, including: notifications, protective equipment, decontamination, activities, and emergency rescue and treatment.
- g. Discuss the general safety precautions necessary for the handling, storage, and disposal of radioactive material.



2. SUMMARY

The fundamentals of radiation protection as related to emergency response is a characteristic function of the radiation itself.

- The type (alpha, beta, gamma, x-ray, neutron)
- The activity in terms of exposure (mR/hr)
- The half-life
- The chemical form (solid, liquid, or gas)
- Can be fixed (nontransferable) or nonfixed (transferable)

To prevent the inhalation or ingestion of radioactive material, the responder must be able to immediately assess any given situation enabling that person to choose and wear the proper protective clothing. Additionally, choice of radiation measuring instruments must be addressed.

A variety of radiations are frequently encountered at DOE facilities. These radiations can be classified into two broad categories--ionizing and nonionizing. **Ionizing** radiations are those radiations that possess sufficient energy to eject electrons from neutral atoms. They include alpha particles, beta particles, gamma rays, x-rays, and neutrons. **Nonionizing** radiations can excite electrons to higher energy states, but do not possess sufficient energy to eject electrons from the atom. Examples of nonionizing radiations (or devices that produce nonionizing radiations) include ultraviolet, visible, infrared, microwave and radio, power frequencies, radar, and lasers.

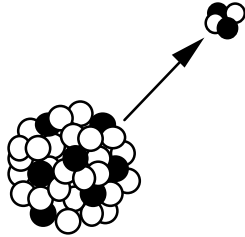
Both ionizing and nonionizing radiations pose potential health hazards in the workplace. Radiological controls and practices should be tailored to the facility and the specific radiation hazard(s). The focus in this competency is to emphasize **ionizing radiations**.

Ionizing radiations can be generated via natural or man-made processes. Natural sources of radiation and radioactivity include cosmic radiations, cosmogenic radionuclides (those radionuclides produced by cosmic ray bombardment with the upper atmosphere), terrestrial radiations, radionuclides in the body, and radon. Man-made sources include medical diagnosis and therapy, consumer products, occupational activities, miscellaneous environmental activities (e.g., air emissions from DOE facilities), and exposures associated with nuclear power generation.

The following figures depict the most commonly encountered ionizing radiations. Note that with the exception of x-rays, each of the radiations is emitted from the nucleus of the atom. X-rays are produced by electronic transitions between shells that surround the nucleus.

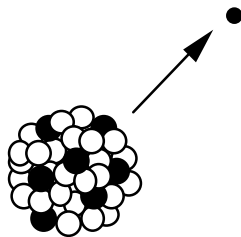


Alpha Particles



- Positively charged helium nuclei.
- Particulate radiations with relatively high energies, but weak penetrating abilities.
- Unlike other ionizing radiations, do not constitute an external hazard.

Beta Particles

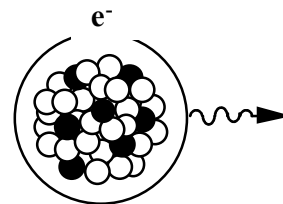
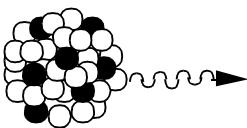


- High-speed electrons formed by the conversion in the nucleus of a neutron into a proton or a proton into a neutron.
- Can be either negatively charged (negatron) or positively charged (positron).
- Particulate radiations with a range in matter greater than an alpha particle.

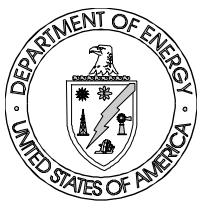
Gamma Rays

and

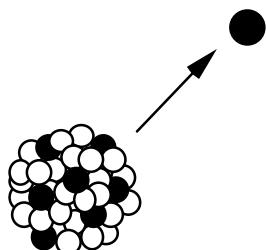
X-Rays



- Electromagnetic radiations with no charge or mass.
- Distinction between gamma rays and x-rays based on origin and energy: gamma rays produced within the nucleus; x-rays outside the nucleus. X-rays, in general, have lower kinetic energies (lower EM frequencies) than gamma rays.
- Very penetrating forms of radiation that travel indefinite distances.



Neutrons



- Particulate radiations with no charge.
- Wide range of energies ranging from thermal (0.025 eV) to fast (several MeV).
- Very penetrating forms of radiation that travel indefinite distances.

Several characteristics of these ionizing radiations are noted in the table below.

Characteristics of Ionizing Radiation

Type	Symbol	Composition	Mass (amu)	Charge	Typical Energies	Range (Air)	Range (Tissue)	Primary Hazard	Examples
Alpha Particle	α	2p + 2n	4	+2	4 - 8 MeV	few centimeters	50 to 70 micrometers	internal	uranium, radon, plutonium
Beta Particle	β	electron	0.00055	± 1	.018 - 3 MeV	up to a few meters	few millimeters	external and internal	strontium-90, carbon-14, tritium
Gamma Ray	γ	electromagnetic ray	0	0	0.1 - 2 MeV	indefinite	indefinite	external and internal	cobalt -60, cesium-137
X-ray	x	electromagnetic ray	0	0	.01 - 150 keV	indefinite	indefinite	external and internal	x-ray machines
Neutron	n	neutron	1	0	0.025 eV - 15 MeV	indefinite	indefinite	external and internal	reactors, neutron sources

Some general points can be made for each of the radiations noted in the table.



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Ionizing radiations constitute both internal and external hazards. For potential internal radiation hazards, the primary objective is to avoid taking in any radioactive materials into the body. This can be realized to a large extent by utilizing **containment** and **confinement techniques** along with cleanliness to minimize the risk of intake through inhalation, ingestion, injection, or open wounds. Adequate protection against excessive exposures to external sources of radiation can be provided by employing three major exposure-reducing principles: **time**, **distance**, and **shielding**. The control of exposure **time** (time spent in a radiation field) is the first major health physics principle available to an occupational worker to limit his/her exposure to an external radiation source. It is important to realize that the radiation dose received by the worker is **directly** proportional to the time spent in a radiation field. Therefore, to minimize the dose received, the time spent in the radiation field must be accordingly reduced. The control of exposure time is a significant factor in the issuance of radiation work permits (RWPs) common at DOE facilities.

A very common and extremely effective technique to reduce personnel exposure is to increase the **distance** between the worker and the radiation source. In many instances, this approach is more important than controlling exposure time and can be easily demonstrated for point (small) sources of radiation. While the exposure-time relationship follows a direct dependence, (i.e., reducing the time spent in a radiation field by one-half reduces the exposure to the worker by one-half), distance dependence often follows the "inverse-square" (second power) law. Thus, doubling the distance from a point source reduces the exposure to the worker by a factor of four. It should be noted that situations do exist where the inverse square law does not apply. In these cases, the relationship between the dose received and the distance from the source does not always follow a simple rule.

A third factor for controlling external exposures entails the use of **shielding**. Shielding the source of radiation becomes important when minimizing time and maximizing distance are not sufficient to reduce personnel exposures to acceptable levels. Determining the required shielding is influenced strongly by the type (alpha, beta, gamma, x-ray, neutron) and energy of the radiation.

Alpha particles have typical energies on the order of 4 to 8 MeV, but rapidly lose this energy through the ionization process. This results in a short range (penetration) in air and tissue and minimal shielding requirements. Beta particles do not ionize to the same degree as alpha radiation. Therefore, they have a greater range than alpha particles, but are still relatively easy to shield. Given the energy of an alpha or beta particle, the range in any medium can be calculated and the appropriate amount of shielding determined.

Because they are uncharged, gamma, x-ray, and neutron radiations are more difficult to shield. The basic approach to gamma and x-ray shielding is to determine what the exposure rate is, and then what it should preferably be after shielding. Calculations of the estimated amount of shielding of a particular material required to reduce the intensity of a beam of gamma or x-ray radiation are then performed. Neutron shielding is based on moderating (slowing down) and thermalizing high energy neutrons, often followed by capture processes. A wide range of shielding materials are used for these purposes.

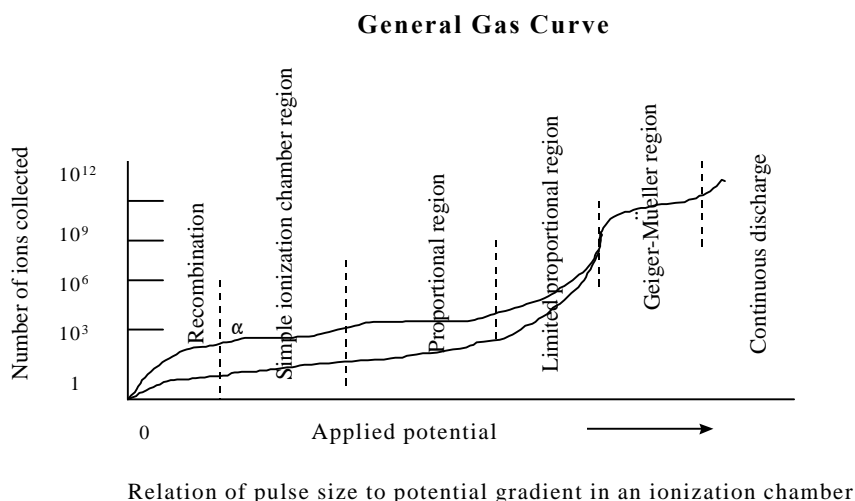


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The following table identifies typical shielding materials for ionizing radiations.

Radiation Type	Shielding Materials
Alpha	Air, paper, dead layer of human tissue
Beta	Plastic, glass, aluminum
Gamma, X-Ray	Lead, tungsten, iron, depleted uranium (in general, materials with high atomic numbers)
Neutron	Paraffin, polyethylene, water, boron, cadmium (in general, materials with low atomic numbers)

The following graphic shows the relative penetrating abilities of alpha, beta, and gamma radiations



Each of the radiations described thus far has the potential to cause a number of biological effects. These effects can be classified into stochastic and nonstochastic (deterministic) effects.

Stochastic effects

- Statistical in nature.
- Probability of the effect occurring within a population increases with dose.
- Generally assumed to have no threshold, implying that even low radiation doses cannot be excluded.
- Include cancer induction and genetic mutations.



Nonstochastic (Deterministic) Effects

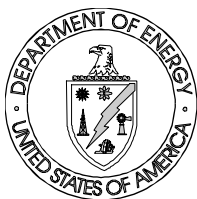
- Severity in an individual varies with the magnitude of the radiation dose.
- The greater the dose, the more severe the effect.
- Assumed to have a threshold radiation value below which the effect is not observed.
- Examples include acute radiation effects observed in individuals exposed to large amounts of radiation, and opacity in the lens of the eye.

The effects of radiation upon biological systems depend primarily upon the total radiation dose and also upon the dose rate (how fast the dose is received). **Acute effects** (early effects) refer to biological effects that occur within one to two months following a radiation dose of approximately 10 rad or more. Inherent in the definition of an acute dose is that the dose was received acutely or promptly (i.e., over a period of up to a few hours). **Chronic** (delayed) effects, as opposed to acute effects, typically occur more than two months, and up to several years, after receiving much smaller doses accumulated steadily on a day-to-day, year-to-year basis.

Ionizing radiation is known to cause biological damage on the cellular level. Radiation is believed to interact primarily with the DNA molecule (deoxyribonucleic acid)--the carrier of genetic information. Radiation produces chemical changes that can eventually lead to cell death or other harmful effects. At low doses and low dose rates, biological repair mechanisms do exist to help counter a radiation "insult;" however, changes occurring on the cellular level can still translate into **carcinogenesis** (cancer induction), **mutagenesis** (genetic defects), and **cell lethality**.

Protecting occupational workers and the public from an elevated cancer risk is the main concern of regulatory agencies. Cancer induction is a stochastic effect that can be observed at low dose rates and over extended periods of time following exposure. Typical environmental levels are well below the threshold values for nonstochastic effects.

Because cancer induction is a stochastic effect, it is assumed to follow a no-threshold relationship with dose. Scientifically, however, this is very difficult to prove and is often debated by health effects experts. Nonetheless, the DOE assumes for regulatory purposes that there is no threshold for the onset of carcinogenesis.



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There are several commonly encountered radiation quantities and units used in the field of radiation protection. The following table serves as a selected summary.

Quantity	Symbol	Units	Radiation Type	Absorbing Medium
Exposure	X	roentgen (R) coulomb/kilogram (C/kg)	gamma, x-rays	air
Absorbed Dose	D	rad gray (Gy) joules/kilogram (J/kg) ergs/gram	any ionizing radiation	any type
Dose Equivalent	H	rem sievert(Sv) joules/kilogram (J/kg) ergs/gram	any ionizing radiation	human tissue (living)

Additional information regarding these quantities and units includes:

Exposure (X)

- Basic concept - Describes an x-ray or gamma ray radiation field. It is a measure of the amount of ionization produced in air by x-rays or gamma rays.
- The conventional unit is the roentgen (R). In the international system (SI) of units, the coulomb/kilogram (C/kg) is substituted for the roentgen. One roentgen = 2.58×10^{-4} C/kg.
- The quantity exposure is only defined in air. It would be incorrect to say, "my dose was one roentgen" because the use of the roentgen indicates that reference is being made to the quantity exposure - a quantity not defined for human tissue.
- This quantity is considered outmoded by the International Commission on Radiation Units and Measurements (ICRU).

Absorbed Dose (D)

- Basic concept - Amount of energy absorbed per unit mass in the medium of interest.
- The conventional and SI units are the rad (no abbreviation) and the gray (Gy), respectively.



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- Unit conversions: $1 \text{ Gy} = 100 \text{ rad}$
 $1 \text{ Gy} = 1 \text{ joule/kilogram (J/kg)}$
 $1 \text{ rad} = 100 \text{ ergs/gram}$
- The quantity is not limited to photon radiations; it applies to all types of ionizing radiation.
- The quantity is not restricted to air, but is applicable to all types of material (air, water, human tissue, etc.).

Dose Equivalent (H)

- Basic concept - Has no precise or exact meaning. It is an administrative concept used for the purposes of radiation protection and is subject to change. It is only meant to apply at those doses commonly encountered in the field of radiation protection (in other words, it does not apply to large acute doses and accident situations). It is related to the amount of biological damage to a person from a given dose of radiation.
- The conventional unit is the rem (no abbreviation); the SI unit is the sievert (Sv).
- Unit conversions: $1 \text{ Sv} = 100 \text{ rem}$
 $1 \text{ Sv} = 1 \text{ J/kg}$
- The quantity applies to all types of ionizing radiation.
- The quantity only applies to living humans.
- The dose equivalent (H) is the product of the absorbed dose (D) and the quality factor (Q); therefore, $H = D \times Q$.

Quality Factor (Q)

The quality factor (Q) relates the absorbed dose received by a worker to the dose equivalent. It only applies to chronic, low-level doses. Consider two individuals who receive the same absorbed dose (one worker from gamma rays, the other from neutrons). The biological damage (or risk) will be greater from the neutron dose. Regulatory controls are put in place to limit the risk, and some means must be used to take into account the different risks associated with different types of radiation. The quality factor is used for this purpose. Each type of radiation is assigned a quality factor based upon its potential to cause biological damage. The absorbed dose can then be multiplied by Q to calculate the dose equivalent.



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Typical quality factors for various radiation types are shown in the following table.

Radiation Type	Quality Factor
Gamma	1
X-Ray	1
Beta	1
Neutrons ≤ 10 keV	3
Neutrons > 10 keV	10
Alpha	20

Bioaccumulation is the process by which radioactivity is cumulatively absorbed or taken up by a target organ within the body, or even by the total body (depending upon the radionuclide involved) and becomes resident. This residency is determined by the biological half-life of the material (i.e., the time required for one-half of the radioactive material taken into the body [intake]) be eliminated by any of the regular excretory processes and by the effective half-life of the radionuclide (the number of atoms in a sample or the sample activity decreases exponentially with time; therefore, effective half-life is the time that is required for half of the sample to decay). Effective half-life can be shown equationally:

$$A = A_0 e^{-\lambda t}$$

where:

A = current activity of the sample

A₀ = original activity of the sample

λ = decay constant (0.693/half-life [T_{1/2}] of the radionuclide)

t = the time elapsed between the original date of the sample and the current date

To respond to an emergency, one must understand the various types of portable radiation survey instrumentation, radiation detection devices, and monitoring systems. The first category, portable survey instrumentation, consists of four basic types: ionization chambers, Geiger-Müller (G-M) detectors, proportional counters, and scintillation detectors. The first three types are generally categorized as gas-filled detectors since they all employ a fill gas of some type for proper operation. The latter category utilizes a solid medium for the detection of ionizing radiation. Examples of each of these types will be described along with some of their uses and characteristics.



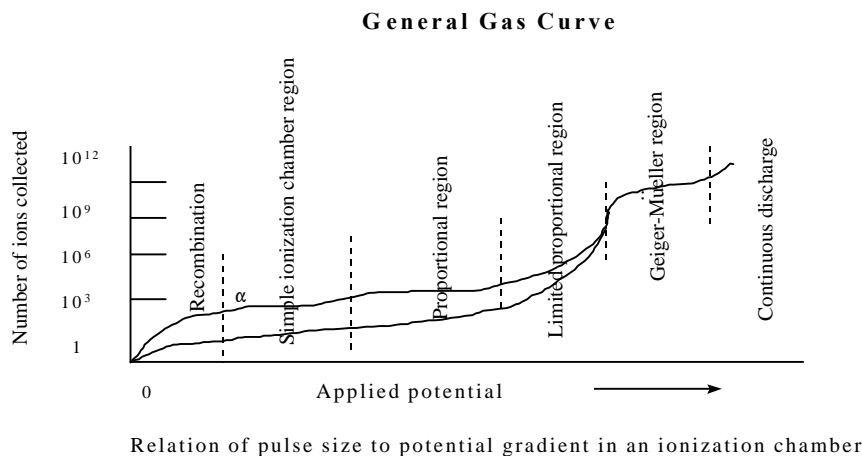
Survey instruments are used for a variety of purposes. Among these are:

- Monitoring contamination levels on equipment and personnel
- Locating lost or hidden sources
- Surveying installations for radiation hazards
- Evaluating the need for posting radiation warning signs
- Predicting the possible exposure in an area and determining the necessity of wearing personnel monitoring devices
- Performing leak tests on radioactive sources

Gas-Filled Detectors

Gas-filled detectors can be designed to detect any of the commonly encountered types of ionizing radiation (alpha, beta, gamma, x-ray, or neutron). In general, these detectors contain a variety of different gases that are either sealed inside a metallic chamber (typically a cylinder) or open to the atmosphere. A wire occupies the center of the chamber. High voltage is supplied to the detector, resulting in the production of an electric field. Radiation interacting within the sensitive volume of the detector is of sufficient energy to "strip" or eject one or more electrons from neutral gas molecules, a process known as gas ionization. The ionization process results in the formation of ion pairs: negatively charged electrons and positively charged gas molecules. The electrons are collected at the central wire either as an electrical pulse (pulse mode) or a current (current mode).

The general gas curve (also referred to as the six-region curve) is a theoretical curve that indicates the general region of operation for gas-filled detectors. The curve appears below.





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In a gas-filled detector, the number of ion pairs measured by the detector per ionizing radiation event is dependent upon the voltage applied to the detector. At low voltages, an ionizing radiation event may not be detected because the ions recombine before reaching the collecting electrode. This area of the general gas curve is referred to as the recombination region.

As the applied voltage increases, ion pairs attain greater kinetic energies and recombination does not occur. This is known as the ionization chamber region. When the voltage is increased above the ionization chamber region, the ions have enough kinetic energy to create new ion pairs after collisions with gas molecules. These new ions are referred to as secondary ions; the number of secondary ions increases proportionally with voltage and with the initial (primary) ions created by the radiation event. This is known as the proportional region. The ratio of the number of secondary ions to primary ions is referred to as the gas amplification factor. As the voltage is further increased, the detector operates in the limited proportional region, a region that is not utilized for radiation detection purposes. Region 5 of the general gas curve is known as the G-M region; any initial ionization event in the detector results in a geiger discharge where the central wire becomes completely saturated with electrons. Region 6 is known as continuous discharge--a region in which certain G-M detectors are rendered inoperative within a short period of time.

From a practical perspective, gas-filled detectors operate in one of three regions: ionization chamber, proportional, or G-M. Each instrument type is designed to operate in one of these regions through the proper choice of construction materials, anode diameters, fill gas, gas-filling pressures, high voltage, etc.

Ionization Chambers

Ionization chambers operate in the ionization chamber region (no. 2) of the general gas curve. Characteristics associated with these detectors include:

- Operate in current mode
- Air is typically used as fill gas
- Gas amplification (multiplication) not required for operation
- Fairly rugged devices
- Short warm-up times (<1 minute)
- Primarily designed to measure x-ray and gamma ray radiations
- Typical readout in units of milliroentgen per hour (mR/hr) or roentgens per hour (R/hr)
- Slow response (relatively insensitive devices)
- Ideal for exposure rate measurements; can measure very high radiation levels with virtually no dead time
- Flat energy response above 100 kiloelectron volts (keV)
- Sensitive to temperature, pressure, and humidity conditions

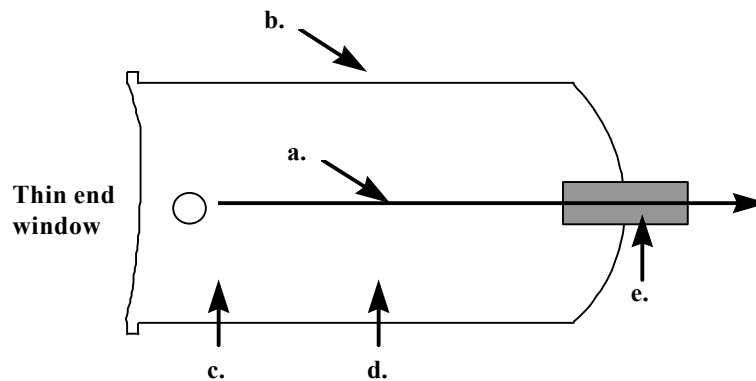


- Detector can "leak" current, most designs require a "zero" strip adjustment
- Can detect/measure alpha and beta radiations with appropriate calibration factors and/or instrument design

Geiger-Müller Detectors

G-M detectors are widely used instruments for the detection of ionizing radiation. These detectors, often referred to as Geiger or G-M counters, are one of the oldest radiation detection devices in existence. A G-M detector, as previously noted, is a gas-filled detector operated in the G-M region (no. 5) of the general gas curve.

All Geiger detectors share certain design features. The diagram below depicts an "end window" G-M tube with key components labeled. A brief description of each follows.



- a. Anode -** A positively charged central wire, typically composed of tungsten, with a diameter on the order of 0.003 to 0.004 inches. Tungsten is favored for its strength and uniformity. The anode is typically a straight wire, but wire loop anodes are found in other designs (e.g., "pancake" detectors).
- b. Cathode -** The outer envelope and conducting surface, negatively charged with respect to the anode. It is usually composed of metal (steel or nickel) and, on occasion, glass that requires an inner conductive coating.
- c. Fill Gas -** A noble gas that occupies 90 to 95% of the active volume of the detector. The noble gas is typically helium, neon, or argon.
- d. Quench Gas -** A gas occupying 5 to 10% of the detector volume. The quench gas functions to prevent the formation of spurious pulses.
- e. Insulator -** Prevents arcing inside the detector.



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High voltage applied to the detector allows the collection of ion pairs--electrons are collected at the anode while positively charged gas molecules migrate to the cathode. The detector usually operates below atmospheric pressure.

To understand the operation of a G-M counter, it might be useful to consider the steps involved in the production of the geiger discharge, that creates pulses of uniform size.

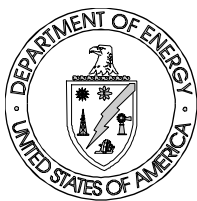
- Step 1: Ionizing radiation enters the detector and strips an electron from a neutral fill gas molecule, creating an ion pair.
- Step 2: Due to the high electric field (high voltage), the "free" electron accelerates toward the anode. As it does so, it acquires sufficient energy to create secondary ionizations. These secondary ionizations serve to dramatically amplify the number of electrons arriving at the anode. This initial amplification is called the Townsend avalanche (the avalanche created by a single original electron).
- Step 3: A series of avalanches follows in rapid succession, propagated by photon emission created by the excitation and subsequent de-excitation of electrons that were not ionized. The wavelength of these photons is in the visible or ultraviolet region.
- Step 4: The anode becomes completely enveloped with electrons indicating a geiger discharge has occurred.

The geiger discharge is formed in approximately one microsecond (μsec) following the initial ionization in the detector. Because the same number of avalanches (statistically speaking) are created each time during this process, the output pulse represents the same amount of collected charge. Therefore, the pulse's height or amplitude remains constant and no energy discrimination is possible.

One of the most significant disadvantages associated with G-M counters is their so-called "dead" time, a period in which the tube does not respond to radiation. It is caused by the slow movement of the positive ions away from the anode; the electric field intensity is too low to produce a geiger discharge.

There are three principal types of G-M counters that are routinely used in health physics:

- end window
- side wall
- pancake



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Principal Type	Description
End Window	The radiation enters the sensitive volume of the detector by passing through a very thin mica window (thinner than a piece of paper) attached to the end of the detector. The window may be protected by a mesh screen. Thin end windows are capable of detecting alpha, beta, and gamma radiations under the appropriate conditions and with proper survey techniques.
Side Wall	This detector has a sliding sleeve that opens and closes from the side. Higher energy beta particles (~300 keV and above) and gamma rays can be detected with the window open; closing the window eliminates the beta contribution and that of lower energy photons.
Pancake	A pancake G-M is similar to the end window in that a very thin mica covering is used. Its design offers a greater detection area than the end window probe in addition to having the same capability of detecting a variety of commonly encountered radiations.

G-M detectors have a wide variety of uses. As a general comment, however, it must be mentioned that because all pulses from a G-M counter are of the same amplitude, no energy discrimination is possible (no spectroscopy). G-M counters do not respond with equal count rates to equal exposures rates from photons of differing energies. Therefore, they are best suited to count rate determinations rather than measurements of exposure, exposure rate, activity, etc. Geiger counters are detection instruments first and foremost. That having been said, some specific applications now follow.

- Contamination Surveys - Fairly rapid monitoring of personnel (hands, clothing, etc.), equipment (tools, etc.), and laboratory surfaces (benches, tabletops, hoods, etc.) can be accomplished using a variety of G-M detectors. When surveying for soft beta emitters, such as carbon-14 (C-14), sulfur-35 (S-35), calcium-45 (Ca-45), and phosphorus-32 (P-32), a thin end window or pancake detector would be required. Higher energy beta and gamma emitters could be detected with end window, pancake, and side wall G-M detectors. These surveys and the detectors involved can be utilized in both laboratory and field applications.
- Leak Testing - Leak testing is a procedure designed to determine whether any removable activity above a specified value is present on the outer surfaces of a sealed source. A smear is taken on the outer surface and counted in a G-M detector. The resulting count rate, with background subtracted, is a measure of the removable activity. If the efficiency of the detector is known, count rates can be converted into disintegration rates for comparison with the guideline value. This procedure is often followed for industrial radiography sources where the opening (port) that the source passes through is smeared with a Q-tip and counted as described above.



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- Accident Dosimetry - Geiger counters can be used for estimating the neutron dose from the activation of sodium-23 (Na-23) to Na-24 in the blood. A "pancake" probe, for example, is placed either against the abdomen of the individual as he/she bends over or under the armpit. Any measurable increase in the count rate (over background) can be an indication of a significant neutron dose. This procedure is referred to as the quick sort method because it can rapidly screen individuals following an accident. The procedure is based on detecting gamma rays emitted from the decay of Na-24.
- Exposure Rate Measurements - In general, measurements of exposure rates can cautiously be performed under two circumstances: when 1) the accuracy of the results is not a crucial concern and 2) the instrument is calibrated for the same energy that will be encountered in the field or laboratory.

A variety of G-M counters can be used for exposure rate measurements (keeping in mind the caveats noted above). These include the typically encountered end window, side wall, and pancake designs. In addition, modified detectors are also available. For example, an energy-compensated side wall G-M tube consists of a rubber sleeve that slides over the tube to flatten the photoelectric response of the detector. Depending on the probe design, exposure rates of up to several R/hr can be measured. A telescoping detector is also available; in this design, a probe containing two halogen-quenched G-M tubes can be extended up to approximately 14 feet from the user and the readout device. Exposure rates of up to 1,000 R/hr can be recorded while the surveyor's dose is dramatically reduced by utilizing distance. This particular G-M detector has practical applications in several areas: radioactive waste surveys, monitoring irradiated fuel storage and transport, monitoring the removal of irradiated samples from reactors, reducing exposure to personnel when locating and evaluating radioactive sources of unknown strength, and emergency radiation accidents.

Typical advantages and disadvantages of GM detectors follow.

Advantages:

- Fairly reliable
- Ease of operation
- Wide variety of shapes and sizes
- Relatively inexpensive
- Highly sensitive (one ion pair can produce a discharge)
- Large output pulses (>1/4 volt to several volts)
- No external amplification normally required due to large amplification factors inherent in the operation of the detector (minimal electronics)



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- Used in field and laboratory settings
- Detect a wide variety of radiations including alpha, beta (soft and hard), x-ray, gamma, and cosmic (high energy gammas)
- Choice of proper operating voltage allows for reproducible results even if the voltage varies
- Excellent for low-level counting rate surveys including personnel and equipment monitoring, leak tests, and as a quick screening method in accident situations
- Halogen tubes have technically infinite lifetimes
- Exposure rate measurements possible under proper conditions

Disadvantages:

- No energy discrimination (spectroscopy is not possible)
- Principally detection, not measurement, devices
- Quenching required to eliminate multiple pulsing
- Worst resolving times of any gas-filled detector
- Slope of the plateau must be kept reasonably flat for reproducible results
- Organic tubes have limited lifetimes
- Self-absorption in the counter wall and window is possible for alpha and beta radiations
- Efficiency is quite poor for gamma rays (approximately 1%)
- Without antisaturation circuits, detector can saturate in high radiation fields and read lower than the true value or even "zero"

Proportional Counters

Proportional counters are extremely versatile instruments used for the detection of ionizing radiation. They share certain design features. Various key components are described below.

Anode - Typically composed of tungsten, with a diameter of approximately 0.001 inches. The anode either takes the form of a loop or straight wire. The nature of gas amplification in a proportional counter requires an extremely uniform central wire.

Cathode - The outer envelope and conducting surface, negatively charged with respect to the anode, and usually composed of steel.

Fill Gas - The gas that occupies the sensitive volume of the detector. It may be an inert gas (argon, krypton, xenon) or a hydrocarbon (methane, ethylene). Other gases are used depending on the application. A very common proportional gas, known as P-10, consists of a mixture of 90% argon and 10% methane. The methane serves as a quenching agent.

Insulator - Prevents arcing inside the detector.



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A proportional counter operates in the proportional region (region 3) of the general gas curve where the applied high voltage is sufficiently high to create secondary ionizations. In contrast to G-M counters, where all pulses are of the same amplitude, the size of the pulse in a proportional counter is proportional to the initial number of ion pairs produced in the detector volume.

When ionizing radiation enters the sensitive volume of a proportional counter, ion pairs are created. The free electrons that are initially produced accelerate toward the anode; secondary ionizations result from the potential applied to the detector. This is known as gas amplification. The number of electrons that arrive at the anode constitute an avalanche (Townsend avalanche). In these respects, proportional counters are similar to G-M counters. Here the similarities end, however. G-M counters operate with amplification factors on the order of one billion (10^9); a series of avalanches eventually envelopes the entire anode, producing pulses of uniform size. In contrast, proportional counters rely on much lower amplification; values in the one thousand (10^3) to one hundred thousand (10^5) range are typical. The anode does not become saturated with electrons and the pulse height is proportional to the initial number of electrons produced in the gas. Energy discrimination with the ability to distinguish radiations becomes possible.

Proportional counters are known for their short dead times. These counters have the capability to distinguish two pulses (two separate ionizing events) in a short period of time. Since each avalanche is restricted to a short section of the anode, unlike G-M counters, the counter can clear this avalanche and respond to a new ionizing event in a time frame approximating 0.5 to 5 μ sec. This is a decided advantage when high counting rates are involved.

A variety of counters operating in the proportional mode exist for routine and specialized applications. Two of the more common examples are:

- Air Proportional - These counters respond only to alpha radiation. Alpha particles enter the detector through a thin window of aluminized mylar. The fill gas is air instead of a noble gas. These lightweight, portable counters are useful for contamination surveys, but must be used with caution in areas of high humidity. For this reason, they are most often encountered in the western half of the United States, where humidities are lower and the response of the detector is not adversely affected.
- Gas Flow Proportional (field use) - These portable instruments respond to both alpha and beta radiation through the appropriate selection of operating voltage. In one design, the fill gas (often liquid propane) is contained in a small canister inside the instrument housing; gas is fed from the canister through a teflon tube housed inside an electrical cable to the probe. The canister is usually replaced every four to six hours. Other designs utilize larger as the source of the counting gas (often P-10). The counter can be purged of air and operated as a stand-alone unit if desired.



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Common applications associated with proportional counters include:

- Contamination Surveys - Detection of alpha and beta radiations can be performed with portable instruments (air proportional and gas flow counters). Large floor surfaces, for example, can be rapidly screened for contamination using a portable, multiwire anode gas-flow counter with a 600 cm² effective surface area. The floor monitor can be moved over the area of interest to quickly identify contaminated locations. Follow-up surface contamination measurements can then be performed with other field proportional counters or another instrument of choice.
- Neutron Detection - Detection of slow neutrons can be accomplished using pulse height discrimination. In a very common reaction, boron trifluoride gas interacts with slow neutrons to produce alpha particles. The alphas are counted while gamma rays are rejected based on their respective pulse heights. The neutrons are detected indirectly by the formation of alpha particles. Other proportional gases are routinely used to accomplish the same objective. The counter can also be modified through the use of moderators to detect fast neutrons.
- Assay of Alpha, Beta, and X-ray Sources - Proportional counters can be used to assay (measure) source activities under the proper conditions. The thickness of the source and source backing must be considered in terms of absorption, especially for alpha and beta radiations.

Typical advantages and disadvantages of proportional counters follow.

Advantages:

- Versatile instruments (wide variety of applications)
- Variety of shapes and sizes available
- Highly sensitive (counter can respond to the formation of one ion pair)
- Size of pulse proportional to initial number of ion pairs
- Can detect (directly or indirectly) a variety of radiations: alpha, beta, gamma, x-ray, and neutrons
- Can distinguish radiations (alpha, beta, etc.) based on pulse height discrimination
- Energy discrimination (spectroscopy)
- Ability to count at much higher rates, relative to G-M counter, because of excellent resolving times (0.5 to 5 μ sec)
- Not only detection, but measurements of dose and dose equivalent possible
- Used in field and laboratory setting



Disadvantages:

- Stable high voltage required due to nature of gas amplification
- External amplification (preamplifiers, amplifiers) required to produce pulse of sufficient size for detection
- Generally more expensive than G-M counters
- Proper operation requires more attention on the part of the user
- Instruments tend to be "finicky" (i.e., more attention to maintenance is required [not as reliable as geiger counters])
- Susceptible to environmental conditions (heat, humidity)
- Self-absorption possible in counters employing entrance windows
- Efficiencies are poor for higher energy x-rays and gamma rays

Scintillation Detectors

In contrast to gas-filled detectors (Geiger counters, proportional counters, and ionization chambers), a solid medium can be used as the sensitive volume for the detection of ionizing radiation. The use of a solid in this regard can be found in the case of inorganic (noncarbon) scintillation detectors. The scintillation mechanism (described below) was the first method ever used to detect ionizing radiation having been observed by Roentgen as a fluorescence on a screen during his discovery of x-rays. Rutherford's classical scattering experiments with alpha particles also relied on the scintillation process. Many years later, solid scintillators of various types are widely used for the detection of alpha (α^{++}), beta (β^{-}), x-ray (x), gamma ray (γ), neutron (n), and proton (p^{+}) radiations.

The process by which an electrical pulse is generated consists of four main steps:

- Step 1: Interaction of ionizing radiation with the detector producing electron-hole pairs.
- Step 2: Conversion of the energy deposited in the detector produced into a proportional amount of light.
- Step 3: Conversion of the light emitted by the scintillator into photoelectrons at the photocathode of the photomultiplier tube.
- Step 4: Multiplication of the initial number of photoelectrons into a measurable electrical pulse.

The end result is that an electrical pulse is produced whose amplitude is proportional to the energy deposited in the scintillator by the incident radiation.



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The following are examples of two commonly encountered inorganic scintillation detectors.

- Sodium Iodide (NaI) - An alkali halide whose discovery and use dates back to the late 1940s. Thallium (Tl) is added in trace amounts as an activator or "wavelength shifter" in order to produce a wavelength of light preferably in the visible region or a wavelength that closely matches the spectral sensitivity of a photomultiplier tube.

NaI(Tl) has several notable characteristics. The crystal exhibits an excellent light yield and can be machined into a variety of shapes and sizes. Negative characteristics include its fragility (sensitivity to thermal and mechanical shocks), and its hygroscopicity (propensity for absorbing moisture). Also, the scintillation photon is not always emitted as a prompt fluorescence--a disadvantage for applications involving high counting rates. Lastly, NaI detectors are energy-dependent devices and should typically be calibrated to the energy of interest.

- Zinc Sulfide (ZnS) - Lord Rutherford used ZnS to visually observe alpha particle interactions during his early scattering experiments. This inorganic scintillator has an efficiency comparable to NaI(Tl). Unlike sodium iodide, however, zinc sulfide is available only as a crystalline powder with density thicknesses on the order of 25 mg/cm². Silver is used as the doping agent. Its use is limited to thin films or screens for alpha and other heavy ion (protons, for example) detection. In practice, ionizing radiation (typically alpha) penetrates a thin aluminized mylar covering prior to interacting with the ZnS(Ag) powder. This covering can be easily punctured resulting in pinhole (or worse) leaks. The detector will then respond to extraneous light sources.

Radioactive contamination is the presence of radioactive material in an unwanted place. Radioactive contamination can occur as:

- Fixed surface contamination
- Removable surface contamination
- Airborne contamination or any combination of the three

Fixed contamination can not be readily removed from a surface. It cannot be removed by casual contact; however, it may be released if the surface is disturbed by activities such as buffing, grinding, sanding, or cleaning with a volatile solvent. Over time, fixed contamination may leach from the material containing it and become removable contamination.

Removable or transferable contamination is radioactive material that can be easily removed from a surface. It may be transferred by any casual contact such as touching, wiping, or brushing. Radioactive material suspended in the air is referred to as airborne contamination and it can be caused



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by occurrences such as air movement over removable contamination, a leak from a radioactive system, grinding on a surface with fixed or removable contamination, or a fire involving radioactive material.

Engineering controls, administrative controls and personnel contamination control practices are used to control radioactive contamination. Engineering controls consist of structures or components designed to minimize or prevent the release of radioactive materials. Ventilation systems and containment systems are two examples of engineered controls. In general, ventilation systems are designed to move any potential airborne contamination away from personnel and to filter the radioactive material from the air. High Efficiency Particulate Air (HEPA) filters, personnel air locks, downdraft tables, hoods, and air movers are examples of ventilation system components that are used to control contamination. Containments are used to physically enclose radioactive material, thereby preventing it from becoming contamination. Containments include items such as vessels, pipes, cells, gloveboxes, glovebags, tents, huts, and plastic coverings.

Administrative controls are used to direct the actions of personnel in order to minimize the risk of exposure to radioactive contamination. Examples of administrative controls include postings to control access to contaminated areas and procedures to minimize or prevent the production of radioactive contamination.

Personnel contamination control practices are measures taken to use personal protective equipment to prevent personnel from becoming contaminated with radioactive material and they include:

- The use of precautionary clothing
- Anticontamination clothing (Anti-Cs)
- Respiratory equipment

Surface contamination is monitored for by radiological control technicians (RCTs) who are required to routinely survey areas for potential contamination. Contamination monitoring equipment is primarily used to detect alpha particles; however, monitoring for beta particles and gamma rays also occurs. Surface contamination may be detected either by a direct frisk (holding the probe of a radiation detection instrument just above the monitored surface) or by smearing or swiping the surface with a paper or cloth swipe and monitoring the swipe for contamination. Direct frisking will detect fixed or removable contamination, while smearing will only detect removable contamination. When surface contamination is detected using the direct frisk method, a smear of the contaminated surface is taken to determine if the contamination is fixed or removable.

Airborne contamination is detected by specialized air monitoring equipment. Airhead samplers contain filters and are connected to vacuum systems so that the air being sampled will flow through the filter while any particulate airborne radioactive contamination present will deposit on the filter.



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The filters are periodically removed and measured for radioactive contamination. Airhead samplers produce very accurate measurements of airborne contamination; however, they do not provide a timely indication of airborne contamination to warn workers. A Selective Alpha Air Monitor (SAAM) is a type of continuous air monitor used to measure airborne radioactive contamination and warn workers if the level rises above the alarm setpoint. Portable air samplers may also be used to measure airborne contamination if permanently installed SAAMs do not provide adequate coverage during maintenance or other activities.

Radioactive contamination can enter the body in four ways:

- Inhalation - breathing in radioactive material.
- Ingestion - eating or drinking radioactive material.
- Injection - radioactive material enters the body through an open wound or as a result of bodily injury such as a puncture wound.
- Absorption - radioactive material absorbed through the skin.

Internal contamination can be prevented by the proper use of contamination control practices and, in particular, the use of engineering controls to isolate radioactive material from personnel. In some instances however, personal protective equipment must be used as the last line of defense against internal contamination. In these cases, respiratory equipment (respirators, supplied air, Self Contained Breathing Apparatus [SCBA], etc.) with an adequate protection factor can be used to prevent the inhalation or ingestion of radioactive material. Additionally, administrative controls to prohibit eating, drinking, smoking, and chewing tobacco may be used to minimize the risk of ingestion of radioactive materials.

The risk of internal contamination by injection can be reduced by minimizing operations where a sharp contaminated object could pierce a containment and injure personnel. Additionally, medical personnel are required to evaluate/decontaminate personnel wounds/cuts that occur to workers while they are in a radiological buffer area, airborne radioactivity area, Contamination Area or High Contamination Area. The risk of internal contamination by absorption can be reduced by the appropriate use of personal protective clothing and prompt decontamination of the skin.

Bioassay samples, lung counting, and wound counting are three methods used in monitoring for internal contamination. A routine bioassay program is required by 10 CFR 835 for personnel if there is a likelihood that an intake of radioactive material could occur which would exceed specific limits as specified in §835.402 (c). Bioassay programs use urine samples, fecal samples, and nasal and mouth smears to determine the committed effective dose equivalent (CEDE) from internal contamination. Fecal samples are the most sensitive bioassay method for detecting internal plutonium contamination.



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The type of bioassay samples collected and the CEDE will depend upon many factors such as the radionuclide involved, the chemical form of the material, the route of entry, and the elapsed time since the intake occurred.

Lung counters are used to detect inhaled radioactive material in the lungs. The lung counter has a limited capacity for detecting internal contamination because it can only measure gamma rays. The amount of inhaled plutonium (Pu) must be estimated from the measured amount of gamma rays emitted from americium (Pu decay product). Lung counts can be used in conjunction with bioassay samples to determine CEDE.

Wound counters measure gamma rays and x-rays emitted by radionuclides deposited in a wound. Wound counts are primarily used as a tool to determine if internal contamination of a wound has occurred and to determine the successfulness of subsequent decontamination efforts.

In general, personal protective equipment used for radiological purposes can be divided into two main types: protection against internal contamination and protection against external contamination or sources. Respiratory protection is used to protect personnel from inhaling or ingesting radioactive material. Respirators (with the appropriate cartridge), supplied breathing air and SCBA are examples of respiratory equipment used to prevent internal contamination. Protection factors are assigned to respiratory protection equipment, and are used to determine the maximum level of airborne radioactive contamination against which the equipment can protect. Examples of equipment used to protect against external contamination or radiation sources include: Anti-Cs (protection against skin contamination), safety glasses (protection against beta to the eye), lead aprons (protection against gamma to reproductive organs), and lead-lined gloves (protection against skin contamination and gamma to hands). Requirements to use personal protective equipment are included in radiological work permits (RWPs) and on postings at area entry points.

According to DOE Order 151.1, *Comprehensive Emergency Management System*, during the response phase of an operational emergency, initial notifications shall be made to workers, emergency response personnel, and organizations, including DOE elements and other local, state, tribal, and Federal organizations. The manager/administrator of each DOE- or contractor-operated site/facility shall notify state and local officials and the DOE Field and Headquarters Emergency Operations Centers within 15 minutes and all other organizations within 30 minutes of the declaration of an Alert, Site Area Emergency, or General Emergency.



3. SELF-STUDY SCENARIOS/ACTIVITIES AND SOLUTIONS

Scenario 1, Part A

On May 31, two workers employed at a DOE contractor facility were tasked with installing a new process line in an indoor building posted and controlled as a high-radiation and high-contamination area. This activity was infrequently performed. Both workers had completed Radiological Worker I training and additional training to allow them access into high-radiation areas. Both workers were currently in compliance with 10 CFR 835 training requirements. However, one worker (Worker "A") required retraining effective the first day of the following month. The workers had been issued and had signed a Radiation Work Permit (RWP) limiting the scope of work to installing the new line in a shielded area of the building. The RWP required a full set of protective clothing without respiratory protection based on the scope and location of the work. Personnel dosimetry requirements consisted of a pocket ionization chamber (0 to 200 mR scale) and a thermoluminescent dosimeter (TLD) badge.

The workers entered the area and began installing new pipe. Operations continued smoothly until late in the afternoon when the workers discovered an out-of-service drain line interfering with installation of the new line. Unfortunately, they failed to observe a faded "Caution: Radioactive Materials" posting placed on the drain line. Because of the time, they decided to quit for the day.

The following morning, the workers informed their supervisor of the situation. The supervisor determined that work could not continue until a flanged pipe tee, connected to the drain line, was removed. Worker "A" attempted to remove the pipe tee, but, having difficulty loosening it, asked for assistance from Worker "B". After five minutes and considerable effort, the tee was successfully removed. Worker "B" observed that one of his gloves had been badly torn during this process, so he removed it and left it on the floor. He then spent a couple of minutes closely examining, touching, and measuring the end of the drain in order to locate a cap that would fit the exposed opening. Not finding an appropriate match, he decided to leave the end open. The two workers spent the following ten minutes one foot away from the old drain line while connecting another section of the new process line. After installation was completed, the workers departed the work area, removed their protective clothing, and performed whole-body frisking. Worker "A" was free of contamination; Worker "B" found contamination on his hands. A radiological control technician (RCT) was notified.

List several concerns raised in this scenario.



Scenario 1, Part B

Following decontamination of Worker "B's" hand, the RCT performed a survey near the drain line. His instrumentation indicated a whole-body dose equivalent rate of 60 mrem/hr at a distance of 30 centimeters. The RCT observed that the open end of the drain line contained an unknown residue. Taking adequate precautions, he collected samples from the drain line; isotopic analyzes performed immediately after collection revealed the presence of plutonium-238 (Pu-238) and plutonium-239 (Pu-239) in the nitrate form. Because of the potential and concern for internal deposition of radioactive material, urine and fecal samples from both workers were obtained for the next several days. Results for Worker "A" were negative. Bioassay results for Worker "B" indicated an intake of 10 Bq of Pu-238 and 12 Bq of Pu-239.

1. What are some concerns raised in this scenario?
2. Estimate the whole body external dose equivalent received by Workers "A" and "B" due to exposure from the out-of-service drain line only.

NOTE: To aid you in your calculation, assume that the workers maintained a constant one-foot distance from the drain line and:

- each worker initially spent five minutes at the drain line discussing what to do about the pipe tee obstruction interfering with their work.
- each worker spent five minutes attempting to remove the flanged pipe tee.
- Worker "B" spent an additional two minutes examining the exposed drain opening.
- each worker spent ten minutes next to the drain line connecting another section of the new process line.

The equation to calculate the external dose equivalent (H) is:

$$H = \text{dose equivalent rate} \times \text{time}$$

3. The simplest way to calculate the committed effective dose equivalent (CEDE) to an individual is to use the tables of exposure to dose conversion factors for inhalation or ingestion found in *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report No. 11 (EPA 520/1-88-020). The CEDE to an individual is calculated:

$$\text{CEDE} = (\text{Intake}) (\text{Dose Conversion Factor})$$



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Another way to calculate the CEDE to an individual is to use a ratio of the intake to the stochastic ALI (sALI). The sALI for a radionuclide is that amount of the nuclide, which if taken into the body by the specified route (inhalation or ingestion), would result in a CEDE of 5 rem. Thus, the CEDE may be calculated:

$$CEDE = \frac{I}{sALI} \times 5 \text{ rem}$$

where:

I = intake

The simplest way to calculate a committed dose equivalent (CDE) to an organ or tissue is to use the tables of exposure to dose conversion factors for inhalation or ingestion found in EPA Federal Guidance Report No. 11. The CDE to an individual organ or tissue is calculated:

$$CDE = (\text{Intake}) (\text{Dose Conversion Factor})$$

Another way to calculate the CDE to an organ or tissue receiving the largest dose is to use a ratio of the intake to the nonstochastic ALI (nALI). The nALI for a radionuclide is that amount of the nuclide, which, if taken into the body by the specified route (inhalation or ingestion), would result in a CDE of 50 rem to an individual organ or tissue. Thus, the CDE to the organ or tissue for which the nALI is specified may be calculated:

$$CDE = \frac{I}{nALI} \times 50 \text{ rem}$$

where:

I = intake

Utilizing the simple way, use the information and equations provided below to calculate the CEDE to Worker "B" and the CDE from these intakes.

Table of Dose Conversion Factors (DCF)			
Radionuclide	Class	CDE per Unit Intake (Sv/Bq)	
-----	-----	Bone Surfaces	Effective
Pu-238	W	1.90E-3*	1.06E-4*
Pu-239	W	2.11E-3*	1.16E-4*

* Taken from EPA Federal Guidance Report No. 11, p. 151


$$H_{50,E} = (Intake)(E DCF)$$
$$H_{50,BS} = (Intake)(E DCF)$$
[illegible]



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Scenario 1, Part A Solution

The scenario as presented raises several initial concerns. These include:

- Lack of a prejob briefing.
- Inadequate administrative control. The significance of the change in the scope of the work went unnoticed. The RWP limited work activities to installing a new drain line only.
- Assigning either worker to this task, considering the entrance to the building was posted as a high-radiation and high-contamination area. A higher level of training is typically recommended (see discussion from DOE 10 CFR 835 and the *Radiological Control Manual* below) for entry into these areas. In addition, Worker "A" required retraining. Even if this individual had the requisite training for entry into these areas, he should not have received authorization to reenter the area on the first day of the month.
- Failure of the work supervisor and the workers to adequately investigate and communicate the situation. While the workers notified their supervisor of the drain line obstruction, there is no indication that: (a) the RWP was reviewed to confirm the scope of the work, (b) the supervisor visually inspected the area, (c) the possibility of external radiation exposure or internal contamination from the pipe was discussed, or (d) a radiological control technician was notified to survey the drain line before and after the pipe tee was removed.
- The faded radioactive materials posting, which, if observed by the workers, could have alerted them and conceivably led to a minimization of the dose received.
- A lack of contamination control. The pipe tee was removed without the benefit of respiratory protection. Worker "B" also tore a glove and made no attempt to discard the glove in an appropriate manner, perform a contamination survey, and replace the glove.
- Failure to perform radiation surveys in a controlled area, resulting in potentially higher exposures to the workers.
- The lack of health physics surveillance. There is no indication that health physics personnel were present in the building (or that portion of the building) to observe and curtail the operation if warranted.

10 CFR 835 contains several subparts and sections relevant to this scenario. Some of the pertinent requirements are noted below.



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Subpart E (Monitoring in the Workplace)

Sections 835.402 and 835.404 address individual monitoring and radioactive contamination control and monitoring, respectively. Individual monitoring requirements essentially center around the use of personnel dosimetry. Section 835.402 requires personnel dosimetric devices for exposures to external radiation. Each worker presumably wore a pocket ionization chamber and a TLD badge based on the RWP requirements. The apparent lack of proper contamination control monitoring (by Worker "B" in particular) is a violation of 835.404.

Subparts F and G (Entry Control Program and Posting and Labeling)

Sections 835.501 and 502 address entries into radiological areas and high/very high radiation areas. Section 835.603 discusses posting requirements for radiological areas. The scenario as presented indicates that posting of the area was performed. Insufficient information exists as to whether all posting requirements and elements of the entry control program were addressed.

Subpart J (Radiation Safety Training)

Section 835.902 is devoted to radiation safety training for radiological workers. This section states requirements for training and retraining at intervals not to exceed two years. In the above scenario, retraining requirements were violated by Worker "A." This section also requires training be commensurate with each worker's assignment. Since the workers were entering a posted high-radiation and high-contamination area, a higher level of training is inferred for these conditions.

DOE/EH-0256T (Revision 1), *Radiological Control Manual*, contains numerous statements that are applicable to this scenario and the concerns noted above. For example:

- Articles 122 and 123 address worker attitudes and responsibilities, respectively. The scenario offers some indication that proper respect for radiation and the responsibilities each worker has when dealing with radiation and radioactive materials needs to be reinforced.
- Article 125 discusses the conduct of radiological operations and recommends that a supervisor be "knowledgeable and inquisitive," ask questions regarding the scope of work, and assist in the development of appropriate procedures. In this case, the supervisor should have requested more information from the workers and considered undertaking a visual inspection of the work location.
- Article 126 notes that properly trained workers can perform "supplementary radiological surveys" when a radiological control technician is not present. These workers apparently did not have any radiological instrumentation with them and, as a result, did not perform surveys of any kind.



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- Articles 221 and 338 advocate frisking when leaving contaminated areas. Worker "B" performed this well enough to detect the presence of contamination on his hands.
- Articles 321 and 322 provide typical information that should be included on and the rationale for using an RWP, respectively. Article 324 offers insight into relevant components of a prejob briefing.
- Article 313 discusses the attention and planning that should be promoted for infrequent or first-time operations. Included in this would be an ALARA review by an appropriate committee and increased line and management oversight. It is conceivable that additional pre-job planning might have limited the worker's exposure.
- Article 334 addresses the minimum recommendations for unescorted entry into a high radiation area. Four criteria should be met: completion of Rad Worker II training (with one exception noted in Article 632.5), training in the use of a survey meter, signatures on the RWP, and the use of personnel and supplemental dosimetry. Note that the two workers had completed Rad Worker I and additional training for access into high-radiation areas. This additional training satisfies the first condition of Article 334. Both workers had signed the RWP. The workers had presumably been trained in the use of a survey meter, but no survey instruments were carried into the area and no surveys were ever performed. The workers carried personnel dosimetry, but no supplemental dosimetry.

NOTE: Some consideration could conceivably be given to the fact that even though the door to the building was posted as a high-radiation and high-contamination area, the work took place in a part of the building where a radiation area existed. The workers did meet the requirements for work in a radiation area. Even so, Worker "A" should not have been allowed access on the following day.

- Recommendations for unescorted access into high-contamination areas include Radiological Worker II training (no exceptions are given), signatures on the RWP, protective clothing and respiratory protection when specified on the RWP, prejob briefings, and personnel dosimetry. Examining these five recommendations, the workers should not have been allowed access to the building because they had not completed Rad Worker II training. As mentioned previously, no prejob briefing had occurred.
- Articles 631-633 discuss the Radiological Worker Training requirements for access to radiological areas.



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- Article 641 advocates that training not only stress normal or routine operations, but also situations where radiological conditions change during the course of performing a particular work function. Dose rates, for example, could increase as the job proceeds, underscoring the importance of recognizing, evaluating, and anticipating changing conditions that could affect a worker's exposure. Training requirements for radiological control technicians and supervisors are specified in Articles 642-644.

Scenario 1, Part B Solution

1. DOE 10 CFR 835 and the *Radiological Control Manual* address some of the concerns in this part of the scenario.
 - The faded radiological posting present on the drain line is a concern. According to 10 CFR 835, Section 601, signs shall be "clear and conspicuously posted." Article 231 of the *Radiological Control Manual* states that postings should "alert personnel to the presence of radiation and radioactive materials," "be conspicuously posted and clearly worded," and "be maintained in a legible condition." The workers' failure to observe the posting is clearly not entirely their fault, but likely resulted in Worker "B" receiving a higher dose.
 - The reading of 60 mrem/hr at 30 cm qualifies as a radiation area under 10 CFR 835 Subpart A, Section 835.2 and as noted in Table 2-3 of the DOE *Radiological Control Manual*. Posting the drain line as a radiation area should have been performed under Article 234.
 - 10 CFR 835.402 requires monitoring in the workplace for exposures to internal radiation. Articles 136 and 361 from the *Radiological Control Manual* refer to the difficulty in measuring transuranic uptakes. For that reason, considerable attention should be paid to controlling and preventing internal exposures. Article 316 cites the need for appropriate engineering and administrative controls as primary and secondary methods, respectively, to limit internal exposures. Respiratory protection is the next resort. Because: (1) respiratory protection was not required on the RWP based on the original scope of work (no potential for airborne radioactivity was thought to exist), and (2) the significance in the change in job scope was not recognized by the workers or the work supervisor, respiratory protection was not utilized at the time the pipe obstruction was discovered, removed, and opened. As a result, one of the workers received an internal dose.
 - Annual allowable dose limits are provided in Subpart C, Section 202 of 10 CFR 835 and Article 213 of the *Radiological Control Manual*. While the whole body and organ limits were not exceeded in this case, the doses received by the workers were not maintained ALARA.



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2. Calculating the external whole-body dose equivalent received by the workers can only be estimated in this case because there are uncertainties regarding: 1) general exposure rates in the shielded portion of the building where they were working (no information was provided), and 2) the workers' proximity to the drain line at any given time. A constant one-foot distance was chosen to simplify the calculation. Given these uncertainties, the whole body doses are estimated as follows:

Worker "A"

Worker "A" spent an estimated 20 minutes near the drain line. Therefore, the worker received a dose equivalent of:

$$(60 \text{ mrem/hr}) \times (1 \text{ hr}/60 \text{ minutes}) \times 20 \text{ minutes} = 20 \text{ mrem}$$

Worker "B"

Worker "B" spent an additional two minutes near the drain line. The dose equivalent is:

$$(60 \text{ mrem/hr}) \times (1 \text{ hr}/60 \text{ minutes}) \times 22 \text{ minutes} = 22 \text{ mrem}$$

3. The CEDE is calculated as follows:

$$\begin{aligned} H_{50,E} &= [(10 \text{ Bq} \times 1.06 \text{ E-4 Sv/Bq}) + (12 \text{ Bq} \times 1.16 \text{ E-4})] \\ &= 2.45 \text{ E-3 Sv} = 0.245 \text{ rem (245 mrem)} \end{aligned}$$

The CDE is calculated as follows:

$$\begin{aligned} H_{50,B} &= [(10 \text{ Bq} \times 1.90 \text{ E-3 Sv/Bq}) + (12 \text{ Bq} \times 2.11 \text{ E-3 Sv/Bq})] \\ &= 4.4 \text{ E-2 Sv} = \mathbf{4.4 \text{ rem}} \end{aligned}$$



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4. Considering only the dose received from exposure to the drain pipe, neither the DOE annual whole-body limit of 5 rem (from both internal and external radiation) nor the organ/tissue dose limit of 50 rem was exceeded for either worker. A summary of the doses received by both workers is summarized in the table below.

SUMMARY OF DOSES RECEIVED BY WORKERS			
Worker	External Dose (mrem)	Internal Dose (mrem)	TEDE
A	20	-----	20
B	22	245	267

Worker "A" received an external dose of 20 mrem and no internal dose for a total effective dose equivalent (TEDE) of 20 mrem, while Worker "B" received a TEDE of 267 mrem (245 mrem internal and 22 mrem external). No information was provided in the scenario as to whether any facility administrative control limits (ACLs) were exceeded. The ALARA philosophy suffered in this instance, however; therefore, an ALARA review should be initiated to prevent this situation in the future.

Scenario 2, Solution

1. The major hazard associated with x-ray diffraction units is the intense, localized exposure from the primary beam to the hands or the eyes that can occur during a change of samples or beam alignment. This scenario involves a situation where the sample could not be enclosed in a protective structure and a "shutter" was left open, exposing one of the worker's hands. (**NOTE:** This cannot occur in many of the new, interlocked units.)
2. The primary beam is very small, but can result in intense fields on the order of several hundred thousand roentgen per minute (R/min). These exposure rates can produce severe dermatological injury and potential loss of fingers.



4. SUGGESTED ADDITIONAL READINGS AND/OR COURSES

Readings

- Argonne National Laboratory. (1988). *Department of Energy Operational Health Physics Training* (ANL-88-26). Argonne, IL: Author.
- Cember, Herman.. (1996). *Introduction to Health Physics* (3rd ed.). McGraw-Hill: New York.
- Gollnick, Daniel A. (1991). *Basic Radiation Protection Technology* (2nd ed.). Pacific Radiation Corporation: Altadena, CA.
- International Commission on Radiological Protection. (1977). *Recommendations of the International Commission on Radiological Protection* (ICRP 26). New York: Author.
- International Commission on Radiological Protection. (1991). *Recommendations of the International Commission on Radiological Protection* (ICRP 60). New York: Author.
- National Council on Radiation Protection and Measurements. (1993). *Limitation of Exposure to Ionizing Radiation* (NCRP Report No. 116). Bethesda, MD: Author.
- National Research Council, National Academy Press. (1990). *Health Effects of Exposure to Low Levels of Ionizing Radiation* (BEIR V Report). Washington, DC. Author.
- United Nations Scientific Committee on the Effects of Atomic Radiation. (1988). *Sources, Effects, and Risks of Ionizing Radiation* (UNSCEAR 1988 Report to the General Assembly). New York: Author.
- U.S. Environmental Protection Agency. (1987). *Radiation Protection Guidance to Federal Agencies for Occupational Exposure* (52 FR 2822). Washington, DC. Author.
- U.S. Environmental Protection Agency. (1988). *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report No. 11 (EPA 520/1-88-020). Washington, DC, and Oak Ridge, TN: Author.

Courses

- *Nuclear Physics/Radiation Monitoring* -- DOE.
- DOE/EH-0450 (Revision 0), *Radiological Assessors Training (for Auditors and Inspectors) - Fundamental Radiological Control*, sponsored by the Office of Defense Programs, DOE.
- DOE/EH-0450 (Revision 0), *Radiological Assessors Training (for Auditors and Inspectors) - Applied Radiological Control*, sponsored by the Office of Defense Programs, DOE
- *Applied Health Physics* -- Oak Ridge Institute for Science and Education.
- *Radiation Protection Functional Area Qualification Standard Training* -- GTS Duratek.



Emergency Management Competency 1.3

Other

Actions or situations were combined to create new incidents from the following references:

- DOE/EH-0450 (Revision 0), *Radiological Assessors Training (for Auditors and Inspectors) - Applied Radiological Control, Lesson 12-i (Radiation-Generating Devices)*. {NOTE: This reference is from a course sponsored by Defense Programs at DOE.
- U.S. Department of Energy. (1996). *Operating Experience Weekly Summary. (96-17, April 19-25, 1996. Final Report No. 1.)*. Washington, DC, Office of Nuclear and Facility Safety.